NON-PUBLIC?: N
ACCESSION #: 8802170085

LICENSEE EVENT REPORT (LER)

FACILITY NAME: North Anna Power Station, Unit 1 PAGE: 1 of 4

DOCKET NUMBER: 05000338

TITLE: Automatic Reactor Trip Due to Hi-Hi Steam Generator Level
EVENT DATE: 01/13/88 LER #: 88-005-00 REPORT DATE: 02/10/88

OPERATING MODE: 1 POWER LEVEL: 015

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: E. Wayne Harrell, Station Manager TELEPHONE #: 703-894-5151

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: At 0313 hours on January 13, 1988, Unit 1 automatically tripped from approximately 15 percent power (Mode 1). The initiating signal for this Reactor Trip was a turbine solenoid trip which resulted when a Hi-Hi level (greater than 75 percent) was detected on 2 out of 3 level channels in the "B" Steam Generator (S/G). The Hi-Hi level in the "B" S/G resulted from S/G level oscillations due to the low temperature of the FW (approximately 81 degrees F) entering the S/G's. This event is reportable pursuant to 10CFR 50.73(a)(2)(iv).

When a Hi-Hi level was reached in the "B" S/G, a turbine solenoid trip occurred which resulted in a turbine trip, reactor trip, and a Main Feedwater (FW) isolation. As a result of the Main FW isolation, the auxiliary FW system automatically started, as designed. As corrective actions, Main FW was restored and Auxiliary FW was secured and then restored to the automatic position. To prevent recurrence of this type event, Operating Procedure 2.1, Unit Power Operation Mode 2 to Mode 1, will be revised to include a caution about FW temperature and increasing power above ten percent with S/G level oscillations.

This event posed no significant safety implications because all safety related equipment responded as designed and key reactor parameters stabilized following the reactor trip. The health and safety of the public were not affected.

(End of Abstract)

1.0 Description of Event

At 0313 hours on January 13, 1988, Unit 1 automatically tripped from approximately 15 percent power (Mode 1). The initiating signal for this Reactor Trip was a turbine solenoid trip above P-7 (2 out of 4 power range nuclear instruments greater than 10 percent power or 1 out of 2 turbine impulse chamber pressure greater than 10 percent rated turbine power). The turbine solenoid trip resulted when a Hi-Hi level (greater than 75 percent) was detected on 2 out of 3 level channels in the "B" Steam Generator (S/G) (EIIS System Identifier JB, Component Identifier SG). This event is reportable pursuant to 10CFR 50.73(a)(2)(iv). A four hour report was made in accordance with 10CFR50.72(b)(2)(ii).

Prior to the ractor trip, Unit 1 was commencing a power rampup, following a chemistry hold, from 5 percent power. Subsequent S/G level control was difficult due to the unusually cold Feedwater (EIIS System Identifier JB) being fed into the S/G's. When S/G level was believed to be adequately controlled, reactor power was increased above 10 percent and the Unit was placed on line. Almost immediately, reactor Tave dropped from approximatley 556 degrees F to 551 degrees F. In response to this condition, operators closed the steam dump valves, but rapid and severe level oscillations occurred on all three S/G's.

When a Hi-Hi level was reached in the "B" S/G a turbine solenoid trip (which resulted in a reactor trip because power was above the P-7 setpoint) and a Main Feedwater (FW) isolation occurred. As a result of the Main FW isolation, the Auxiliary FW pumps started, as designed. At approximately 0320 hours, Main FW was restored and Auxiliary FW was secured and then returned to the automatic position.

Following the reactor trip, primary system pressure and temperature decreased to approximately 2140 psig and 533 degrees F, respectively, then recovered to the normal no load values of 2235 psig and 547 degrees F. The Reactor Coolant System (RCS) temperature normally decreases to 547 degrees F following a reactor trip. The Reactor Coolant System (RCS) temperature decrease to 533 degrees F was due to the negligible decay heat in the core, the time it took to manually isolate the Moisture Separator Reheater (EIIS System Identifier SB, Component Identifier RHTR) warmup lines, and the isolation of the Main FW which required operation of the Auxiliary FW system. The relatively long running time of the steam driven auxiliary FW pump contributed to the cool down because the steam used to drive the pump was being extracted from the S/G's. The decrease in RCS temperature resulted in letdown isolation when pressurizer level decreased to approximately 14.5 percent on one channel.

Plant equipment responded as expected for seven Individual Rod Position Indicators (IRPI) (EIIS System Identifier AA, Component Identifier ZI), which did not indicate correctly. The IRPI for control rod D-10 indicate approximatley 32 steps and its rod bottom light did not illuminate. The IRPI's of the other six control rods indicated that these rods were slightly withdrawn; however, the rod bottom lights properly indicated that these rods were inserted. All seven IRPI's were subsequently recalibrated, and this confirmed that all control rods fully inserted following the reactor trip signal.

2.0 Safety Consequences and Implications

This event posed no significant safety implications because all

safety related equipment responded as designed and key reactor parameters stabilized following the reactor trip. The health and safety of the public were not affected.

3.0 Cause of the Event

The cause of the reactor trip was a Hi-Hi level in the "B" S/G which resulted from S/G level oscillations. The root cause of the S/G level oscillations was attributed to very cold FW (approximately 81 degrees F) beind fed into the S/G's as a result of a low condenser hotwell temperature (approximately 55 degrees F). The low hotwel temperature was due to the severe cold weather and the fact that the normal flow path for returning auxiliary steam and heating steam drips to the condenser was isolated to enhance meeting the secondary system water chemistry specifications required for operation.

4.0 Immediate Corrective Action

As an immediate corrective aciton, Main FW was restored and Auxiliary FW was secured and then restored to the automatic position.

5.0 Additional Corrective Action

As an additional corrective action, the warmup lines on the MSR's were isolated and auxiliary steam was lined up to the hotwell to raise the temperature of the condensate water.

6.0 Actions Taken to Prevent Recurrence

To prevent recurrence of this type event, Operating Procedure 2.1, Unit Power Operation Mode 2 to Mode 1, will be revised to include a caution about FW temperature and increasing power above ten percent with S/G level oscillation..

7.0 Similar Events

On September 28, 1984, a turbine trip was manually initiated, on Unit 1 at North Anna Power Station, in anticipation of a Hi-Hi S/G level. The turbine trip resulted in a reactor trip and was reported under LER 84-014-00.

8.0 Additional Information

Unit 2 was stable in Mode 1 throughout this event and was not affected.

Subsequent to this trip, Unit 1 was taken to Mode 2 and then to Mode 5 in order to facilitate cleanup of the secondary system and to perform other desired maintenance.

10 CFR 50.73

VEPCO VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION
P.O. BOX 402
MINERAL, VIRGINIA 23117

February 10, 1988

U.S. Nuclear Regulatory Commission Serial No. N-88-004
Attention: Document Control Desk NO/DEQ: nih
Washington, D.C. 20555 Docket No. 50-338

License No. NPF-4
Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 1.

Report No. LER 88-005-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to Safety Evaluation and Control for their review.

Very Truly Yours,
/s/
E. Wayne Harrell
Station Manager

Enclosure

cc: U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

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